UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

September 25, 1985

Docket No. 50-395

Mr. O. W. Dixon, Jr.
Vice President Nuclear Operations
South Carolina Electric & Gas Company
P.O. Box 764
Columbia, South Carolina 29218

Dear Mr. Dixon:

Subject: Issuance of Amendment No. 45 to Facility Operating License NPF-12 Virgil C. Summer Nuclear Station, Unit No. 1

The Nuclear Regulatory Commission has issued Amendment No. 45 to Facility Operating License NPF-12 for the Virgil C. Summer Nuclear Station, Unit No. 1 located in Fairfield County, South Carolina. This amendment is in response to your letter dated March 6, 1985, and supplemented April 30, 1985, and August 9, 1985.

The amendment modifies the Technical Specifications to reflect a 1.9% reduction in thermal design flow. The amendment is effective seven days after its date of issuance.

A copy of the related safety evaluation supporting Amendment No. 45 to Facility Operating License NPF-12 is enclosed.

Notice of issuance will be included in the Commission's next monthly Federal Register notice.

Sincerely,

for Elinor G. Adensam, Chief
Licensing Branch No. 4
Division of Licensing

Enclosures:
1. Amendment No. 45
2. Safety Evaluation

cc w/enclosures:
See next page

September 25, 1985

AMENDMENT NO. 45 TO FACILITY OPERATING LICENSE NO. NPF-12 - Virgil C. Summer Unit 1

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Docket No. 50-395
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SOUTH CAROLINA ELECTRIC & GAS COMPANY

SOUTH CAROLINA PUBLIC SERVICE AUTHORITY

DOCKET NO. 50-395

VIRGIL C. SUMMER NUCLEAR STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 45
License No. NPF-12

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment to the Virgil C. Summer Nuclear Station, Unit No. 1 (the facility) Facility Operating License No. NPF-12 filed by the South Carolina Electric & Gas Company acting for itself and South Carolina Public Service Authority (the licensees), dated March 6, 1985, and supplemented April 30 and August 9, 1985, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's regulations as set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, as amended, the provisions of the Act, and the regulations of the Commission;

   C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

   D. The issuance of this license amendment will not be inimical to the common defense and security or to the health and safety of the public;

   E. The issuance of this license amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is hereby amended by page changes to the Technical Specifications as indicated in the attachments to this license amendment and paragraph 2.C(2) of Facility Operating License No. NPF-12 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 45 , are hereby incorporated into this license. South Carolina Electric & Gas Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective seven days after its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

for Elinor G. Adensam, Chief
Licensing Branch No. 4
Division of Licensing

Enclosure:
Technical Specification Changes

Date of Issuance: September 25, 1985

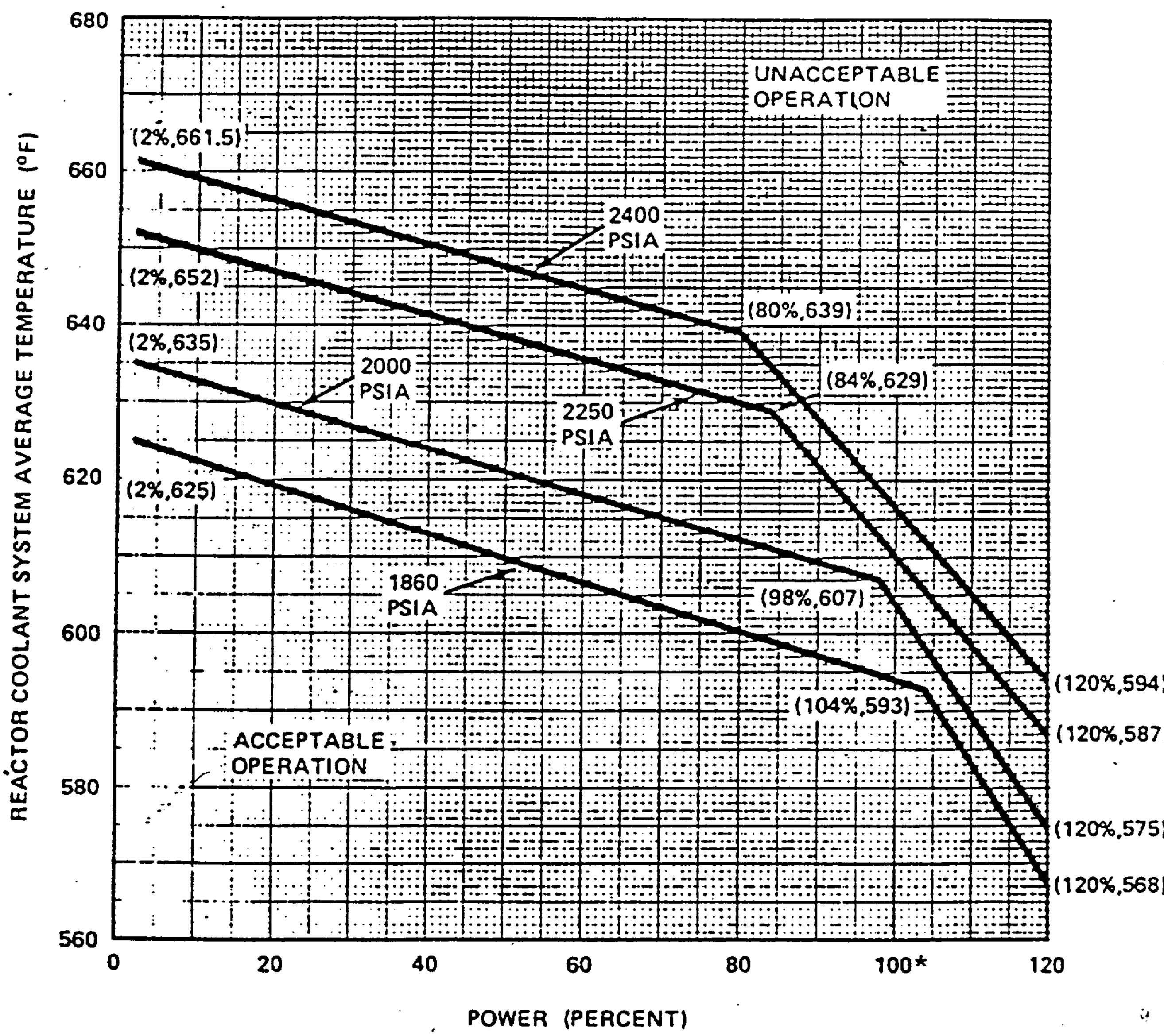
ATTACHMENT TO LICENSE AMENDMENT NO. 45

FACILITY OPERATING LICENSE NO. NPF-12

DOCKET NO. 50-395

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by Amendment number and contain vertical lines indicating the areas of change. The corresponding overleaf page is also provided to maintain document completeness.

Amended Page	Overleaf Page
2-2	
2-5	
3/4 2-8	3/4 2-7
3/4 2-9	
3/4 2-10	
3/4 2-11	
B 3/4 2-4	
B 3/4 2-5	



*When operating in the reduced RTP region of Technical Specification 3.2.3 (Figure 3.2-3), the restricted power level must be considered 100% RTP for this figure.

Figure 2.1-1 Reactor Core Safety Limit - Three Loops in Operation

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

Functional Unit	Total Allowance (TA)	Z	S	Trip Setpoint	Allowable Value
1. Manual Reactor Trip	Not Applicable	NA	NA	NA	NA
2. Power Range, Neutron Flux High Setpoint	7.5	4.56	0	$\leq$109% of RTP	$\leq$111.2% of RTP
Low Setpoint	8.3	4.56	0	$\leq$25% of RTP	$\leq$27.2% of RTP
3. Power Range, Neutron Flux High Positive Rate	1.6	0.5	0	$\leq$5% of RTP with a time constant $\geq$2 seconds	$\leq$6.3% of RTP with a time constant $\geq$2 seconds
4. Power Range, Neutron Flux High Negative Rate	1.6	0.5	0	$\leq$5% of RTP with a time constant $\geq$2 seconds	$\leq$6.3% of RTP with a time constant $\geq$2 seconds
5. Intermediate Range, Neutron Flux	17.0	8.4	0	$\leq$25% of RTP	$\leq$31% of RTP
6. Source Range, Neutron Flux	17.0	10.0	0	$\leq 10^5$ cps	$\leq 1.4 \times 10^5$ cps
7. Overtemperature ΔT	7.1	2.94	1.8	See note 1	See note 2
8. Overpower ΔT	4.5	1.4	1.2	See note 3	See note 4
9. Pressurizer Pressure-Low	3.1	0.71	1.5	$\geq$1870 psig	$\geq$1859 psig
10. Pressurizer Pressure-High	3.1	0.71	1.5	$\leq$2380 psig	$\leq$2391 psig
11. Pressurizer Water Level-High	5.0	2.18	1.5	$\leq$92% of instrument span	$\leq$93.8% of instrument span
12. Loss of Flow	2.5	1.0	1.5	$\geq$90% of loop design flow*	$\geq$89.2% of loop design flow*

*Loop design flow = 96,200 gpm
RTP = RATED THERMAL POWER

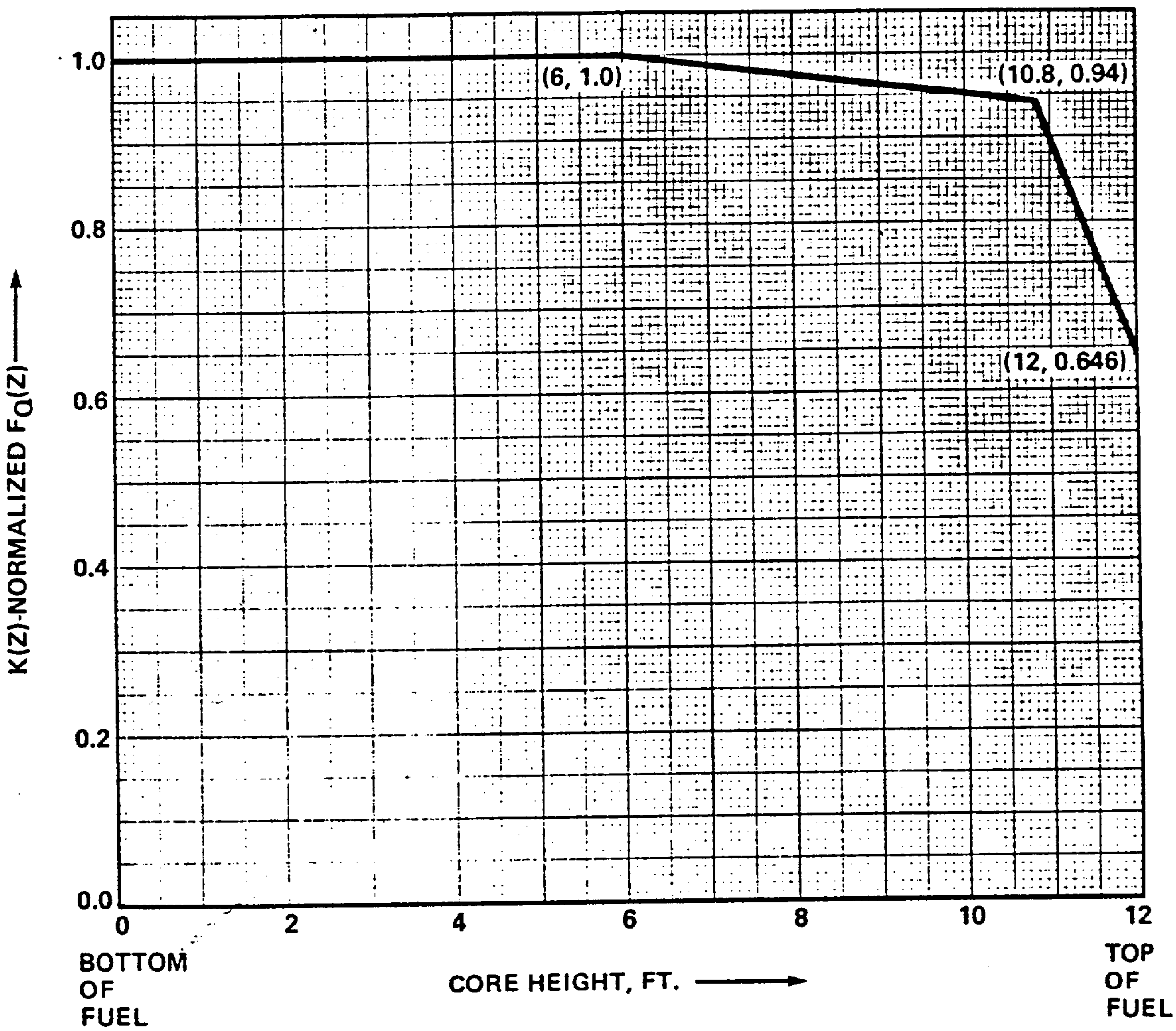


FIGURE 3.2-2

K(Z)-NORMALIZED $F_Q(Z)$ AS A FUNCTION OF CORE HEIGHT

3/4.2.3 RCS FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

LIMITING CONDITION FOR OPERATION

3.2.3 The combination of indicated Reactor Coolant System (RCS) total flow rate and R shall be maintained within the region of allowable operation shown on Figure 3.2-3 for 3 loop operation.

Where:

a. $R = \dfrac{F^N_{\Delta H}}{1.49\,[1.0 + 0.2\,(1.0 - P)]}$,

b. $P = \dfrac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$,

c. $F^N_{\Delta H}$ = Measured values of $F^N_{\Delta H}$ obtained by using the movable incore detectors to obtain a power distribution map. The measured values of $F^N_{\Delta H}$ shall be used to calculate R since Figure 3.2-3 includes measurement uncertainties of 3.5% for flow and 4% for incore measurement of $F^N_{\Delta H}$.

APPLICABILITY: MODE 1.

ACTION:

With the combination of RCS total flow rate and R outside the region of acceptable operation shown on Figure 3.2-3:

a. Within 2 hours either:

1. Restore the combination of RCS total flow rate and R to within the above limits, or

2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER and reduce the Power Range Neutron Flux - High trip setpoint to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.

b. Within 24 hours of initially being outside the above limits, verify through incore flux mapping and RCS total flow rate comparison that the combination of R and RCS total flow rate are restored to within the above limits, or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours.

c. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced THERMAL POWER limit required by ACTION items a.2. and/or b. above; subsequent POWER OPERATION may proceed provided that the combination of R and indicated RCS total flow rate are demonstrated, through incore flux mapping and RCS total flow rate comparison, to be within the region of acceptable operation shown on Figure 3.2-3 prior to exceeding the following THERMAL POWER levels:

ACTION: (Continued)

1. A nominal 50% of RATED THERMAL POWER,

2. A nominal 75% of RATED THERMAL POWER, and

3. Within 24 hours of attaining greater than or equal to 95% of RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.2.3.1 The provisions of Specification 4.0.4 are not applicable.

4.2.3.2 The combination of indicated RCS total flow rate and R shall be determined to be within the region of acceptable operation of Figure 3.2-3:

a. Prior to operation above 75% of RATED THERMAL POWER after each fuel loading, and

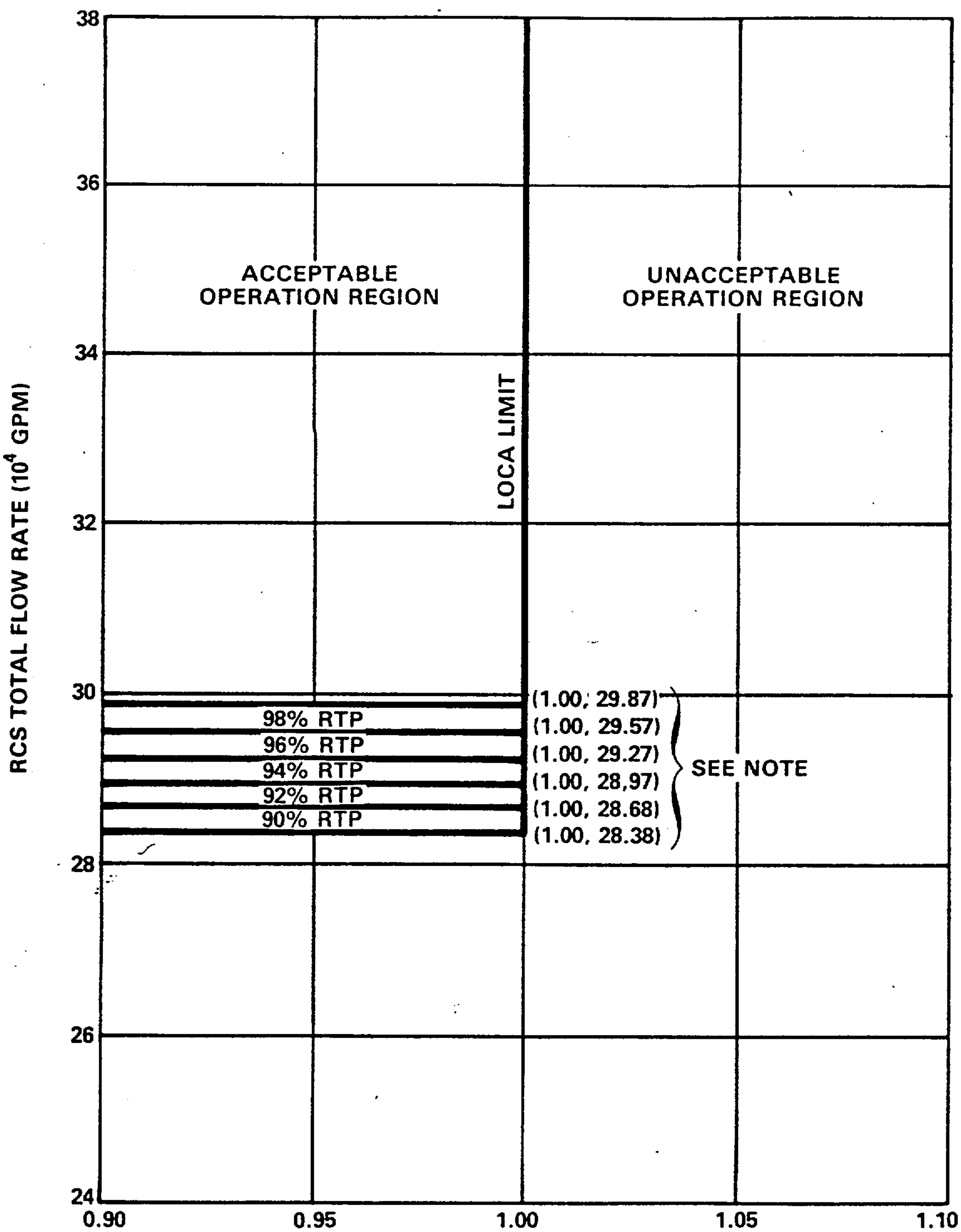
b. At least once per 31 Effective Full Power Days.

4.2.3.3 The indicated RCS total flow rate shall be verified to be within the region of acceptable operation of Figure 3.2-3 at least once per 12 hours when the most recently obtained value of R obtained per Specification 4.2.3.2, is assumed to exist.

4.2.3.4 The RCS total flow rate indicators shall be subjected to a CHANNEL CALIBRATION at least once per 18 months.

4.2.3.5 The RCS total flow rate shall be determined by measurement at least once per 18 months.

MEASUREMENT UNCERTAINTIES OF 3.5% FOR FLOW AND 4.0% FOR INCORE MEASUREMENT OF $F_{\Delta H}^{N}$ ARE INCLUDED IN THIS FIGURE



$$R = F_{\Delta H}^{N}/1.49\left[1.0 + 0.2(1.0 - P)\right]$$

FIGURE 3.2-3 RCS TOTAL FLOW RATE VS. R THREE LOOP OPERATION

NOTE: When operating in this region, the restricted power levels shall be considered to be 100% of rated thermal power (RTP) for Figure 2.1-1.

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HEAT FLUX HOT CHANNEL FACTOR and RCS FLOWRATE and NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

c. The control rod insertion limits of Specifications 3.1.3.5 and 3.1.3.6 are maintained.

d. The axial power distribution, expressed in terms of AXIAL FLUX DIFFERENCE, is maintained within the limits.

$F^N_{\Delta H}$ will be maintained within its limits provided conditions a. through d. above are maintained. As noted on Figure 3.2-3, RCS flow rate and $F^N_{\Delta H}$ may be "traded off" against one another (i.e., a low measured RCS flow rate is acceptable if the measured $F^N_{\Delta H}$ is also low) to ensure that the calculated DNBR will not be below the design DNBR value. The relaxation of $F^N_{\Delta H}$ as a function of THERMAL POWER allows changes in the radial power shape for all permissible rod insertion limits.

R, as calculated in 3.2.3 and used in Figure 3.2.3, accounts for $F^N_{\Delta H}$ less than or equal to 1.49. This value is used in the various accident analyses where $F^N_{\Delta H}$ influences parameters other than DNBR, e.g., peak clad temperature and thus is the maximum "as measured" value allowed.

Fuel rod bowing reduces the value of DNB ratio. Credit is available to offset this reduction in the generic margin. The generic design margins, totaling 9.1% DNBR, completely offset any rod bow penalties.* This margin includes the following:

1) Design limit DNBR of 1.30 vs. 1.28
2) Grid Spacing (K_s) of 0.046 vs. 0.059
3) Thermal Diffusion Coefficient of 0.038 vs. 0.059
4) DNBR Multiplier of 0.86 vs. 0.88
5) Pitch reduction

The applicable value of rod bow penalties is referenced in the FSAR.

When an F_Q measurement is taken, an allowance for both experimental error and manufacturing tolerance must be made. An allowance of 5% is appropriate for a full core map taken with the incore detector flux mapping system and a 3% allowance is appropriate for manufacturing tolerance.

The radial peaking factor $F_{xy}(Z)$ is measured periodically to provide assurance that the hot channel factor, $F_Q(Z)$, remains within its limit. The

*The generic margins also offset the penalty associated with the thermal design flow reduction included in Amendment 45 to the Technical Specifications.

HEAT FLUX HOT CHANNEL FACTOR and RCS FLOWRATE and NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

F_{xy} limit for Rated Thermal Power (F_{xy}^{RTP}) as provided in the Radial Peaking Factor Limit Report per specification 6.9.1.11 was determined from expected power control maneuvers over the full range of burnup conditions in the core.

When RCS flow rate and $F_{\Delta H}^{N}$ are measured, no additional allowances are necessary prior to comparison with the limits of Figure 3.2-3. Measurement errors of 3.5% for RCS total flow rate and 4% for $F_{\Delta H}^{N}$ have been allowed for in determining the limits of Figure 3.2-3.

The 12 hour periodic surveillance of indicated RCS flow is sufficient to detect only flow degradation which could lead to operation outside the acceptable region of operation shown on Figure 3.2-3.

3/4.2.4 QUADRANT POWER TILT RATIO

The quadrant power tilt ratio limit assures that the radial power distribution satisfies the design values used in the power capability analysis. Radial power distribution measurements are made during startup testing and periodically during power operation.

The limit of 1.02, at which corrective action is required, provides DNB and linear heat generation rate protection with x-y plane power tilts. A limiting tilt of 1.025 can be tolerated before the margin for uncertainty in F_Q is depleted. The limit of 1.02 was selected to provide an allowance for the uncertainty associated with the indicated power tilt.

The two hour time allowance for operation with a tilt condition greater than 1.02 but less than 1.09 is provided to allow identification and correction of a dropped or misaligned control rod. In the event such action does not correct the tilt, the margin for uncertainty on F_Q is reinstated by reducing the maximum allowed power by 3 percent for each percent of tilt in excess of 1.0.

For purposes of monitoring QUADRANT POWER TILT RATIO when one excore detector is inoperable, the movable incore detectors are used to confirm that the normalized symmetric power distribution is consistent with the QUADRANT POWER TILT RATIO. The incore detector monitoring is done with a full incore flux map or two sets of 4 symmetric thimbels. These locations are C-8, E-5, E-11, H-3, H-13, L-5, 2-11, N-8.

3/4.2.5 DNB PARAMETERS

The limits on the DNB related parameters assure that each of the parameters are maintained within the normal steady state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the initial FSAR assumptions and have been analytically demonstrated adequate to maintain a minimum DNBR of 1.30 throughout each analyzed transient.

The 12 hour periodic surveillance of these parameters through instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 45 TO FACILITY OPERATING LICENSE NPF-12

SOUTH CAROLINA ELECTRIC & GAS COMPANY

SOUTH CAROLINA PUBLIC SERVICE AUTHORITY

VIRGIL C. SUMMER NUCLEAR STATION, UNIT 1

I. INTRODUCTION

By letter dated March 6, 1985 (Ref. 1), South Carolina Electric and Gas Company submitted a request for an amendment to the Virgil C. Summer Technical Specifications to reflect a thermal design flow reduction of 1.9%. Revised calculations supporting but not changing this request were submitted by letter dated April 30, 1985, (Ref. 2). By letter dated August 9, 1985, (Ref. 3), a note was added to the Technical Specifications bases section to clarify that the available generic design margins are what offset the penalty associated with the thermal design flow reduction of 1.9%. This note did not substantially change the amendment consisting of a thermal design flow reduction of 1.9% as noticed (50 FR 16014) on April 23, 1985, but simply makes clear that the flow reduction is covered by the available margins in the design calculations. Therefore, this amendment request was not renoticed.

II. EVALUATION

LOSS-OF-COOLANT-ACCIDENT ANALYSIS

The licensee provided an evaluation on the effect of reduced design reactor coolant flow on the postulated loss-of-coolant-accident (LOCA). The licensee only analyzed and evaluated double ended cold leg guillotine (DECLG) breaks since these breaks were identified previously as limiting cases that result in the highest peak cladding temperature. The DECLG break analyses were performed with 102% of design thermal power of 2775 MWt and total peaking factor of 2.32. A discharge coefficient of 0.4 was used for the limiting case analysis since the sensitivity study shows that the DECLG break with a discharge coefficient of 0.4 results in the highest peak cladding temperature.

The analyses were performed by using a modified version of the 1981 Westinghouse ECCS evaluation model (Ref. 4). This evaluation model uses the standard PAD Fuel Thermal Safety Model (Ref. 5) for the calculation of the initial fuel rod conditions, the SATAN-VI code for the thermal-hydraulic transient analysis for the RCS during blowdown, the WREFLOOD code for the analysis of the refill and reflood transient period, the COCO code for the containment pressure transient, and the LOCTA-IV code for the calculation of the peak cladding temperature. The modified version of the ECCS evaluation model uses the approved BART code (Ref. 3) to calculate the reflood heat transfer coefficient normally performed by the WREFLOOD code. This code takes no credit for the effects of the grids in increasing reflood heat transfer.

The staff has reviewed the large break LOCA analysis. The calculated peak cladding temperature is 2189.2°F, the maximum local metal water reaction is 5.7% and the total core metal-water reaction is less than 0.3 percent. We, therefore, conclude that the results presented are acceptable since the approved methods and computer codes were used and the analytical results show that the peak cladding temperature, metal-water reaction and clad oxidation are within the acceptance criteria of 10 CFR 50.46.

THERMAL-HYDRAULIC DESIGN

Since the proposed changes to Technical Specifications involve the decrease of thermal design flow of 1.9%, the impact of operating at this lower flow on thermal margin is evaluated.

The licensee has determined that 1.9% flow reduction will result in a DNBR penalty of 3.0%. This is derived from using a previously approved sensitivity factor for the rate of change of DNBR with respect to the flow reduction and is acceptable.

The licensee has also recalculated the rod bow penalty on DNBR by using the approved method (Ref. 6). The maximum calculated rod bow penalty is 2.3% for fuels in the Summer reactor core.

Since the W-3 correlation was used to establish the operating DNBR limit for the Summer reactor core, the generally approved DNBR margin of 9.1% is applicable to the core. This margin is sufficient to compensate for the 2.3% rod bow penalty and penalty of 3.0% DNBR associated with the reduced design flow.

The licensee has also evaluated the impact of the reduced design flow on DNB and non-DNBR related transient responses. As a result of the evaluation the licensee concludes that, even with the design flow reduced by 1.9%, the FSAR conclusion that no safety criteria will be violated during transients remains valid.

Based on our review of the licensee's evaluation process and results, we conclude that the reduction of design flow by 1.9% is acceptable for the transient responses.

TECHNICAL SPECIFICATIONS

The specific Technical Specification changes and the reasons for their acceptability are:

<u>Table 2.2-1</u>

This table has been modified to include the reduction by 1.9% for the loop design flow. This change is supported by the analysis for safe operation of the core and is acceptable.

Figure 3.2-3

This figure has been modified to remove the operating area which has the dependence of operating power and reactor coolant flow on rod bow penalty. The changes cause the operating band to be more restrictive compared with the previously approved operating band and the changes are acceptable.

Changes in pages B3/4.2-4, B3/4.2-5, 3/4.2-8, 3/4.2-9, 2-2 and 3/4.2-11 are editorial. The changes are consistent with the changes related to the removal of dependence of operating band on rod bow penalty (as shown in Figure 3.2-3) and are acceptable.

In conclusion, the staff has reviewed the proposed changes to the Summer Plant Technical Specifications involving a reduction of thermal design flow by 1.9% and finds that they are acceptable.

REFERENCES

1. Letter from O. W. Dixon, Jr. (SCE&G) to H. R. Denton (NRC) dated March 6, 1985.

2. Letter from O. W. Dixon, Jr. (SCE&G) to H. R. Denton (NRC) dated April 30, 1985.

3. Letter from O. W. Dixon, Jr. (SCE&G) to H. R. Denton (NRC) dated August 9, 1985.

4. Letter from C. O. Thomas (NRC) to E. P. Rahe (W), "Acceptance for Referencing of Licensing Topical Report WCAP-9561 - BART A-1: A Computer Code for Best Estimate Analysis of Reflood Transients," dated December 21, 1983.

5. E. P. Rahe, WCAP-9220: Westinghouse ECCS Evaluation Model, 1981 Version, Revision 1, 1981.

6. Letter from J. F. Stolz (NRC) to T. M. Anderson (W), "Review of WCAP-8720, Improved Analytical Models Used in Westinghouse Fuel Rod Design Computations."

7. J. Skaritka, WCAP-8691 (Revision 1), "Fuel Rod Bow Evaluation, dated July 1979.

III. ENVIRONMENTAL CONSIDERATION

This amendment involves a change in the use of a facility component located within the restricted area as defined in 10 CFR Part 20. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR Sec. 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

IV. CONCLUSION

The Commission made a proposed determination that the amendment involves no significant hazards consideration which was published in the Federal Register (50 FR 16014) on April 23, 1985, and consulted with the state of South Carolina. No public comments were received, and the state of South Carolina did not have any comments.

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: Jon B. Hopkins, Licensing Branch No. 4, DL
Summer B. K. Sun, Core Performance Branch, DSI

Dated: September 25, 1985